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Neutron dosimetry of WWER pressure vessels

by

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ABSTRACT

The paper defines the role of neutron dosimetry of WWER pressure vessels in tackling the problems of the radiation resistance of such vessels. The main assumptions of the computational and experimental dosimetry method are examined and the main sources of error in this method are indicated. The status of experimental and computational research on operating nuclear power plants is described and a summary is given of the results of this research. It is pointed out that the problems of monitoring and more accurately determining the neutron fluence incident on pressure vessels can successfully be resolved by out-of-vessel WWER dosimetry.

INTRODUCTION

The radiation embrittlement of water-water power reactor (WWER) pressure vessels is considered one of the main safety problems of nuclear power plants with such reactors. Under the influence of neutrons and other factors, the pressure vessel steel changes its properties during reactor operation. For example, there is a shift to the high-temperature region in the steel brittle fracture threshold at which there is a reduction in ductility. Under certain conditions this can create a danger of brittle fracture of the steel. This applies particularly to pressure vessel welds located in the irradiation zone because of the high content of copper, phosphorus and other elements in the welds. The brittle fracture threshold is characterized by the ductile-to-brittle transition temperature and is denoted by Tk.

There is a danger of brittle fracture of the pressure vessel steel under the following conditions:

(1) Abrupt cooling to the transition temperature of the vessel area subjected to neutron irradiation (occurrence of the so-called

"thermal shock", when there is emergency flooding of the pressure vessel) followed by an increase in pressure;

(2) Prolonged irradiation of the pressure vessel, resulting in a marked increase in Tk in the potential failure area;

(3) The presence of a mechanical defect (crack, cavity, etc.) of sufficiently large (critical) size in the irradiation zone, if there is a possibility of the defect rapidly propagating through the entire thickness of the pressure vessel.

Of course, the presence of these conditions may not lead to vessel failure since fracture mechanics and the criteria for crack propagation in the pressure vessel are probabilistic in nature. Nevertheless, maximum transition temperature values Tk are established for pressure vessel operation, and beyond these the pressure vessel is no longer considered safe.

Neutron irradiation is the principal cause of increases in the transition temperature Tk during operation, Tk being defined as follows:

$$T_K = T_{KO} + \Delta T_N + \Delta T_T + \Delta T_F, \ ^oC \qquad (1)$$

where

Tko is the initial transition temperature;

ΔT_N is the shift in Tk caused by cyclical loading (for the smooth cylindrical part of a WWER pressure vessel this is negligible);

ΔTt is the shift in Tk caused by thermal ageing (for the steel of a WWER pressure vessel this is equal to zero);

ΔT_F is the increase in Tk due to neutron irradiation.

The semi-empirical dependence for the increase ΔT_F is expressed for WWER pressure vessels as:

$$\Delta T_F = A_F \, (\Phi \cdot 10^{-18})^{1/3}, \quad ^oC \qquad (2)$$

where

A_F is the radiation embrittlement coefficient (for example, the value for the base metal of WWER-440 vessels is 9, and for welds - 13);

Φ is the neutron fluence for energies above 0.5 MeV in units $1/cm^2$.

More generally, the increase ΔT_F is a function of many parameters, such as

$$\Delta T_F = f[dpa,\ \varphi(>E),\ C_{Cu},\ C_P, \ldots], \qquad (3)$$

where dpa is the number of displacements per atom, defined as:

$$dpa = \int_0^T\int_0^\infty \sigma_d(E)\varphi(E,t)\,dE\,dt\ , \qquad (4)$$

where

$\sigma_d(E)$ is the neutron cross-section of the atomic displacement in the material at a neutron energy E;

$\varphi(E,t)$ is the energy density of the neutron flux at the place of irradiation as a function of the neutron energy E and time t;

T is the total irradiation time.

In addition:

$\varphi(>E)$ is the integral neutron flux density for neutrons of energy above E;

C_{Cu}, C_P is the concentration of copper and phosphorus impurities in the steel of the base metal or welds.

Clearly, the increase in the transition temperature is largely determined by the increase ΔT_F. For this reason, maximum limits are fixed for the magnitude of the increase $[\Delta T_F]$, which, using dependences (2) and (3), in turn determine the maximum values of the neutron characteristics: fluence - $[\Phi]$ or dpa-[dpa]. The time taken by the reactor pressure vessel to attain these maximum values determines the vessel's safe working life. Because of the importance of knowing the neutron characteristics occurring in WWER vessels in order to assess their safe life and to define dependences (2) and (3), neutron dosimetry is being accorded greater attention. This interest in the study of neutron fields in the pressure vessel area is also due to the technical difficulty of performing measurements at operating nuclear power plants and the labour-intensive nature of the computational and experimental method of determining neutron characteristics. As a result, national research programmes have been set up to examine the behaviour of control samples of the base metal and weld steel irradiated directly in operating plants in positions near the pressure vessel. In addition, experiments are being carried out to determine the neutron characteristics near the pressure vessel in several WWERs, and calculations are being made of neutron transport in the pressure vessel area. The results of the latter are then fitted to the experimental data. Recently, as part of work on the neutron dosimetry of pressure vessels, measurements have been performed in the space behind WWER pressure vessels.

NEUTRON DOSIMETRY OF WWER PRESSURE VESSELS

Having determined the importance of ascertaining neutron characteristics in WWER pressure vessels, let us now focus on the main neutron dosimetry problems being tackled in the research programme on pressure vessel radiation resistance.

1. Monitoring of the neutron fluence at the points where the vessel steel control samples subjected to accelerated irradiation are located inside the reactor vessel. Activation foils are used as accompanying detectors. Capsules containing samples are loaded into the reactor at the start of operation and partially removed for examination after 1, 2, 3 and 5 years of operation during reactor fuelling. The fast neutron fluence incident on the samples will correspond approximately to that on the pressure vessel after 9, 18, 27 and 45 years. However, the difficulty is that the neutron fluence incident on the capsule is distributed unevenly because of the sharp neutron flux gradient at the capsule location and other factors. In addition, the considerable distance between the position of the samples and the pressure vessel makes it difficult to transfer the experimental results for the control samples to the pressure vessel because the neutron characteristics at the vessel and at the locations of the control samples differ substantially. This leads to the second task.

2. The need to determine the pressure vessel neutron characteristics. In this case, the difficulty lies in the fact that the direct measurement of the fast neutron fluence distribution in the pressure vessel and of the differential neutron spectrum is practically impossible. For this reason the computational and experimental method is used to determine the pressure vessel neutron characteristics. The neutron characteristics required are determined with the help of calculations of neutron transport in the space near the pressure vessel. The accuracy of the calculated values is verified using the experimental data obtained from the same space. It is desirable that the experimental studies be conducted as close as possible to the pressure vessel and that they provide information on the distribution of neutron characteristics in terms of both space and neutron energy. For experimental studies at operating WWER plants, it is possible to use the air gap on the outside of the pressure vessel or the ionization chamber channels set into the biological shield. However, if neutron characteristics are being measured directly on the inner surface of the pressure vessel, such experiments are preferred. It should be noted that, in the computational and experimental method of determining neutron characteristics in operating WWER plants, the experiment should precede the calculation. Since the values of the neutron field characteristics in the space near the vessel depend largely on the state of the reactor core at a specific moment of time, it is not possible to model in advance and predict what the operating conditions of a power reactor will be at the time the experiment is to be performed. The neutron source models in the calculation variants should correspond to the state of the reactor core at the time the measurements are made. Only thus is it possible to compare correctly the experimental and computational results. A detailed description of the state and parameters of the reactor core together with a description of the experimental method and the experimental data obtained at the WWER pressure vessel can provide the basis for a benchmark field at the point where the experiment was performed. Using such a field it is possible to verify

the neutron field calculation programs employed in pressure vessel neutron dosimetry and, more specifically, programs for accurate determination of neutron characteristics under conditions of neutron source fluctuation in the reactor core.

3. Monitoring of the actual values of the neutron characteristics and in particular the fast neutron fluence during operation. Design values for the neutron fluence incident on pressure vessels are assessed on the basis of WWER core load and parameters calculated at the reactor design stage, or of experiments at similar types of reactor, or of individual experiments on reactors of this type. However, during operation core loads can differ from design loads because of the more economical arrangement of fuel assemblies in the core. This produces a change in power density in the peripheral part of the core and a corresponding change in the leakage from the core. Similar changes can occur as a result of unforeseen jamming of control elements or a change in the rate of coolant flow through individual loops. These and other factors mean that design values for the neutron fluence impinging on pressure vessels need to be accurately determined towards the end of the operating period. This can only be done if full information on the whole period of operation of the WWER is available. New WWER reactor core compositions have recently been developed and put into use which reduce leakage from the core. This has been done to reduce the fast neutron flux incident on the pressure vessel, which is essential if the degree of embrittlement of pressure vessel steel is to be reduced. In several WWER-440 reactors (1375 MW(th) capacity), 36 edge fuel assemblies were replaced by steel fuel assembly shields. Core compositions with spent fuel assemblies at the edge are considered promising. A number of WWERs function at load with low neutron leakage, and in these spent fuel assemblies having operated for one or two years are placed at the edge instead of the fresh assemblies foreseen in the original plans. In order to achieve a greater reduction in the fast neutron fluxes impinging on the pressure vessel, it is possible to place burnable absorbers and control

Table 1

Fluence reduction needed	Measures
Up to 2	Placing around the core edge fuel assemblies which have been in operation for one or two years instead of fresh fuel assemblies.
2-6	Placing around the core edge fuel assemblies which have been in operation for two or three years, plus installing fuel assembly shields in the place of those fuel assemblies that are closest to the pressure vessel.
Over 6	As above, plus placing control elements in edge fuel assemblies. In addition, a series of measures to prevent brittle fracturing of the pressure vessel: heating the water in the tank of the core emergency cooling system; annealing of the pressure vessel metal; use of the latest monitoring methods to detect defects, etc.

elements in the edge fuel assemblies. Table 1 shows the reduction in the pressure vessel fast neutron fluences that is required and a possible method of achieving this.

Increasing the unit output of WWERs by 5-10% alters the neutron characteristics in the pressure vessel. In order to provide a basis for safe operation of the reactor at higher power during the design lifetime of the pressure vessel, further research on the change in neutron characteristics is needed. A more accurate determination of the neutron fluence incident on the pressure vessel is important also from the point of view of extending the pressure vessel's lifetime beyond design expectations. This requires a thorough analysis of operating regimes and of the change in pressure vessel metal properties using more accurate values for neutron characteristics.

There are two methods of monitoring the actual values of the neutron fluence incident on the pressure vessels of operating WWER plants:

experimental and computational. In the former, neutron monitors are placed in the space near the pressure vessel to follow variations in the pressure vessel neutron flux. With this procedure, the relationship between monitor readings and the neutron fluence at the pressure vessel points in question must first be established. If the control samples are only a very short distance from the pressure vessel, control sample detectors may be used as fluence monitors. However, in this case allowance has to be made for many uncertainties - some of which are referred to above - as well as for the azimuthal uncertainty associated with fluence monitoring. In addition, the complexity and labour-intensive nature of taking measurements make the performance of this type of monitoring throughout the plant's working life a problem. The USSR has developed a different monitoring method using activation detectors positioned just outside the vessel. Figure 1, depicting $^{1}/12$ of a cross-section of a WWER-440 reactor, shows the dimensions and location of the gap in relation to the pressure vessel. The position of this gap in WWER-1000 reactors (3000 MW(th) capacity) is similar. Accessibility when the reactor is shut down, the weak radial gradient of the neutron flux, the possibility of positioning detectors at any point around the reactor vessel both azimuthally and vertically, make the gap a convenient place from which to monitor the actual neutron fluence impinging on the vessel. Also, the detectors can be installed either permanently for every reactor run during the reactor's lifetime or periodically to check calculated neutron flux values. The simplicity of the mounting for attaching monitors to WWER pressure vessels should be noted.

To determine the neutron fluence incident on the pressure vessels of reactors in which no monitors have been installed, the computational monitoring method is used. It is particularly important to perform such determinations for reactors whose design lifetime is coming to an end. In such cases calculations are made of the neutron transport for reactor and source models that are characteristic of each fuel loading. With this method

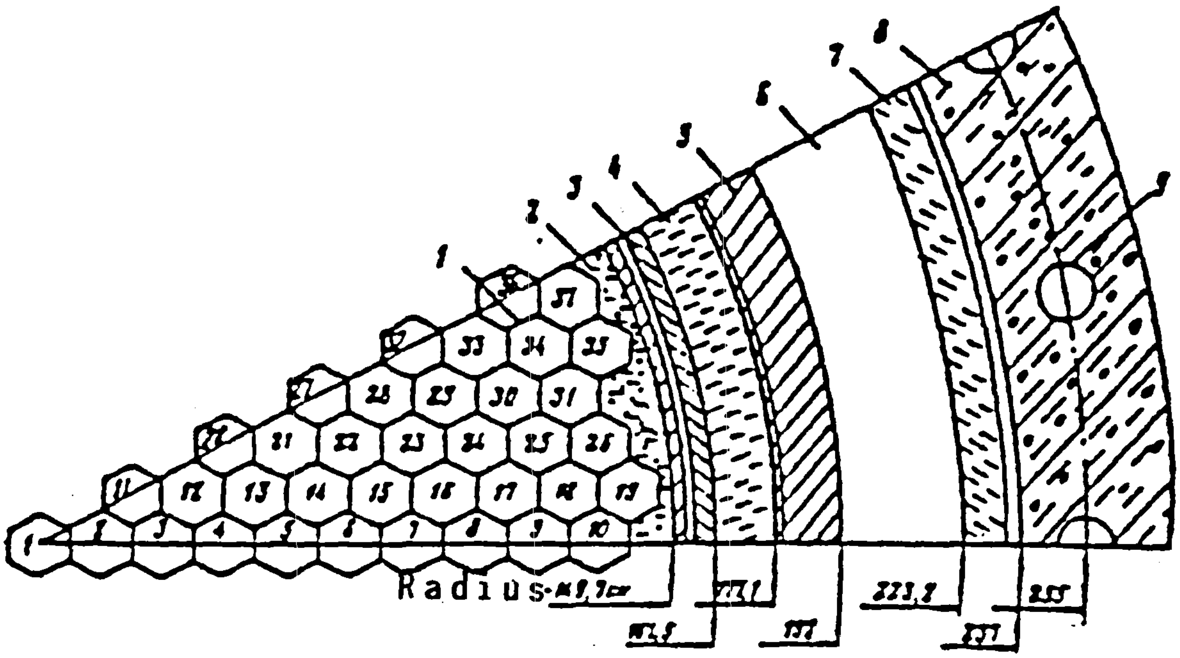

Fig. 1. Diagram of series-produced WWER-40 reactor (30 degree section of the symmetrical sector):

1 - Reactor core
2 - "Hot water"
3 - Cavity
4 - "Cold" water
5 - Vessel with deposition welding
6 - Air clearance
7 - Thermal insulation
8 - Concrete
9 - Ionization chamber channel

uncertainty arises as to the adequacy of the computational model and real structures and core conditions. In the computational method, therefore, detailed information on the composition of the core at a specific moment (or period) of operation is of vital importance, as is the use of multigroup two-dimensional computer codes. For calculation variants it is possible to use an apparatus (developed on the basis of the calculation of conjugate neutron flux density) for the sensitivity of unknown quantities to variation in the initial parameters. It is thus possible to avoid performing many labour-intensive direct-problem calculations for the neutron flux density, although the uncertainties associated with computational modelling are retained. Some of these uncertainties can be partially eliminated by a combination of computational and experimental research, i.e. performing benchmark experiments where this is possible. Where it is not possible to carry out such experiments at operating reactors, the computer programs must be verified on a model (mock-ups) of the pressure vessel structure at a research reactor. This is the fourth problem of neutron dosimetry.

4. The necessity of modelling the WWER structure (a section from the core edge to the pressure vessel and biological shielding) and of undertaking computational and experimental research into neutron characteristics stems from the demand for increased accuracy of calculated values for the neutron flux density and the neutron spectrum in WWER pressure vessels. The possibility of measuring neutron characteristics at practically any point of the mock-up using different methods which provide a high degree of accuracy and the simplicity of geometrical modelling in the structural calculations make the use of these devices standard in studies of the neutron characteristics of WWER pressure vessels using computational methods. They make it possible to verify different neutron transport calculation programs which use various transport equation approximations, and to optimize neutron cross-sections for reactor shielding calculations. Direct measurements and the calculation of the neutron spectrum at the inner surface of the WWER pressure vessel model using methods employed for nuclear power plant reactors make it possible to evaluate the accuracy of results of calculations and to determine the space coefficients for the extrapolation of the neutron flux density from different points in the space near the pressure vessel to the inner surface of the pressure vessel. Modelling of the shielding of WWERs and related studies are being carried out at the LR-0 reactor in Czechoslovakia [1] and at the IR-50 reactor in the Soviet Union [2].

Having determined the main problems of neutron dosimetry of WWER pressure vessels, it is necessary to show the role of neutron dosimetry in defining the safe operating life of pressure vessels. Figure 2 shows a chart of the current assessment of the safe life of WWER pressure vessels using data on neutron characteristics in the pressure vessel (actual fast neutron fluence) and the results of materials tests on the control samples (increase in the transient temperature because of neutron irradiation).

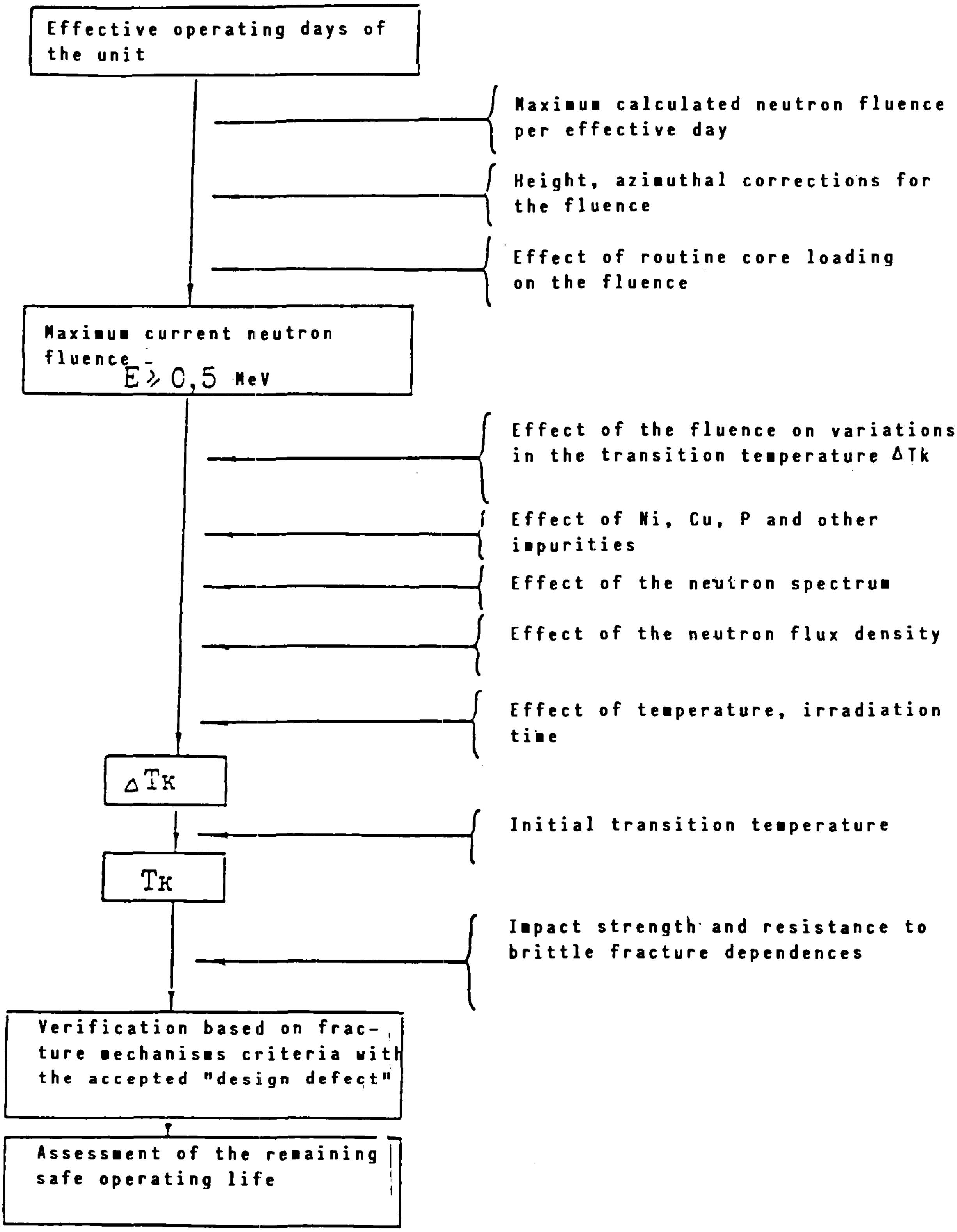

Fig. 2. Diagram of the significance of neutron characteristics in vessel fracture analysis and assessment of the resource in the operational process

In view of the importance of the computational and experimental method in pressure vessel neutron dosimetry, let us describe it in more detail.

THE COMPUTATIONAL AND EXPERIMENTAL METHOD OF NEUTRON DOSIMETRY

1. Calculation of the neutron field in the space near the pressure vessel

The neutron field characteristics of the reactor shielding can be determined by solving the neutron transport kinetic equation. This is done using different numerical methods of solving the transport equation in a multigroup approximation. The traditional way to resolve radiation transport problems is to use discrete ordinates methods, formulated for one-dimensional and two-dimensional geometries. The relative simplicity of the discrete ordinates method algorithm and the possibility of achieving a high level of accuracy in calculations of inhomogeneous shielding geometries have resulted in the creation of programs based on these methods - for example, ROZ-6 and RADUGA, and foreign methods such as ANISN and DOT. These and other programs are widely used to determine neutron characteristics in the shielding of water-water reactors characterized by deeply penetrating radiation and complex structures with inhomogeneities and cavities.

The neutron field calculation is based on the interaction of programs of three main types:

1 - Calculation and preparation of group constants;

2 - Solving a multigroup system of transport equations;

3 - Processing the results obtained.

The group constants are prepared by software systems using neutron data files. For instance, for ANISN and DOT program calculations, constants' libraries DLC-41 (VITAMIN) and DLC-23 (CASK) were prepared, which have 171 and 22 neutron energy groups respectively. For the ROZ-6 and RADUGA programs, a 28-group neutron constants' library was set up.

The type of problems that the ROZ-6 and ANISN programs can solve is one-dimensional problems in plane, cylindrical and spherical geometries. Taking a negligible amount of computer time per calculation, these programs can be employed for estimate calculations and for neutron spectra calculations using multigroup libraries of neutron constants (e.g. DLC-41 in ANISN program calculations).

Programs of the DOT and RADUGA type solve problems in two-dimensional geometry. The calculations, which take into account the cylindrical type of WWERs, are performed in radial-azimuthal (r,θ) and radial-axial (r,z) geometries. However, one calculation variant using these programs consumes a great deal of computer time, and these programs are therefore used for more precise base calculations. The structure of WWERs is such that calculations can be made on a one-twelfth segment of a reactor in r,θ geometry under conditions of angle boundary reflection. A typical segment used for calculations is shown in Fig. 1. It is assumed that the reactor possesses absolute 30-degree symmetry in terms of layout, composition of elements and distribution of neutron sources in the core. If the actual distribution of neutron sources in the core is not symmetrical, the calculation may be made for a 60-degree or 120-degree segment. In calculations in r,z geometry, the core is made cylindrical with an equivalent radius of:

$$R \text{ equiv.} = d\sqrt{\frac{n\cdot\sqrt{3}}{2\pi}} \quad , \qquad (5)$$

where

d is the fuel assembly pitch around the triangle and

n is the number of fuel assemblies.

An important aspect of the calculation of neutron transport in a WWER is the geometrical modelling of the region calculated in the appropriate geometry. Modelling of the core edge in r,θ geometry must be particularly accurate since this is where fuel assembly projections are closest to the pressure vessel. In such places, the subintervals must be as small as possible so as to permit correct modelling of the extreme edge of the fuel assembly. Another important aspect is the correct inclusion of the core neutron sources in the calculation. In order to do this, the first step is to determine the real volume distribution of power density and neutron flux density in the WWER core. Either the experimental or the computational method may be used. The power density distribution in individual fuel assemblies can be measured by means of in-core sensors, which monitor the fuel assembly outlet temperature and measure the height distribution of the neutron flux density by means of direct charge detectors. In view of the fact that not all fuel assemblies contain in-core sensors and that it is impossible to measure the fuel element power density distribution in fuel assemblies in an operating reactor, the computational method is used to determine the neutron sources in the core. With the help of the relevant programs, the power density volume distribution (for both fuel assemblies and fuel elements) can be obtained and is then corrected on the basis of measurement results from individual fuel assemblies. Finally, the volume distribution of the neutron sources is obtained and is then modelled in r,θ or r,z geometries. After the results of the calculations have been processed by two-dimensional or one-dimensional programs, the neutron flux density distribution for the corresponding groups in the space near the pressure vessel are obtained. Let us consider the application of DOT and ANISN programs to calculate the structure of WWERs. In order to obtain a three-dimensional distribution for the neutron flux density $\varphi_g(r,\theta,z)$ of the "g" group, different approximations may be used based on a combination of results of computations in two-dimensional and one-dimensional geometries. For instance, as a result of calculations in r,θ geometry, it is possible to introduce a correction for neutron leakage in the z-direction in the form:

$$k_g(r,z) = \frac{\varphi_g(r,z)}{\varphi_g(r)} , \qquad (6)$$

where

$\varphi_g(r,z)$ is the neutron flux density of the "g" group, obtained from the DOT program calculation;

$\varphi_g(r)$ is the neutron flux density of the "g" group obtained from the ANISN program calculation.

Thus, the expression for the three-dimensional neutron flux density will be:

$$\varphi_g(r,\theta,z) = \varphi_g(r,\theta) \times f_g(r,z) \quad , \qquad (7)$$

where

$\varphi_g(r,\theta)$ is the neutron flux density of the "g" group, obtained from the DOT program calculation in r,θ geometry.

In the computational method developed for WWERs in Ref. [3], which is based on the approximations set forth here, it was noted that equation (7) is applicable over the height of the core.

In conclusion, let us list the principal aspects of the computational method which introduce errors into the results:

1. Approximations of the transport equation calculation;
2. Constants in the neutron energy group division;
3. Geometrical model of the region calculated;
4. Composition of the physical zones of the region calculated;
5. Distribution of the neutron sources in the core;
6. Method of calculating corrections to obtain the three-dimensional neutron flux density.

When comparing computational and experimental data, it is essential to take account of the errors introduced by the above approximations.

2. Experimental determination of neutron fluence and flux density in WWER pressure vessels

Threshold neutron activation detectors employed to determine the characteristics of the fast neutron field acting on the reactor pressure vessel. In WWER plants, threshold detectors are used both in in-core dosimetry experiments (neutron monitors of control samples) and in out-of-vessel dosimetry (measurements in the gap and biological shielding). The activation or fission reaction rate per nucleus of the neutron activation detector is related theoretically to the energy density of the neutron flux φ(E) by the expression:

$$R = \int_0^\infty \sigma(E)\,\varphi(E)\,dE \quad , \qquad (8)$$

where σ(E) is the activation or fission reaction cross-section.

In threshold detectors, the reaction cross-section may be represented as a step function. At a certain energy threshold $E_{eff.}$, the cross-section jumps from zero to a certain value of $\sigma_{eff.}$ and then remains constant when the energy increases further. This method of presenting a cross-section is known as the effective threshold cross-section method and can characterize the threshold reaction of the interaction between the neutrons and the detector nuclei. There are several ways of seeking the effective threshold and of determining the effective cross-section, and these are based on various physical considerations. When the detectors are irradiated in an unknown spectrum, it is recommended that effective threshold and cross-section values from a number of well-researched reactions be used [4]. Evaluating the neutron spectrum at the place where the neutron activation detector is located (for instance, by calculation or by performing measurements on a WWER mock-up at a research reactor), permits the more accurate definition of the effective cross-section. Since in pressure vessel neutron dosimetry the experimental data must be compared with calculated data (obtained by the DOT program, for example), and since the selection of an effective energy threshold is conditional in nature, the effective thresholds of a number of reactions can be taken as equal to the boundaries of the energy groups of the library used in the calculation. The thresholds selected should, of course, be close to, for example, the recommended values. As the detectors used in pressure vessel dosimetry are generally irradiated over a long period of time, the half-life of the reaction product must be comparable with the period of irradiation. Table 2 shows the characteristics of a number of detectors used in the neutron dosimetry of WWER-440 pressure vessels. The effective cross-sections were determined from the following equation:

$$\sigma_{eff} = \frac{\int_{0}^{\infty} \sigma(E)\,\varphi(E)\,dE}{\int_{E_{eff}}^{\infty} \varphi(E)\,dE} \qquad (9)$$

using a 171 group spectrum obtained from a one-dimensional WWER calculation and the detector reaction cross-section library IRDF-82.

Table 2

Reaction	Threshold energy, MeV	Effective cross-section, mb	Product half-life
$^{58}Ni(n,p)$	2.35	370	70.78 days
$^{54}Fe(n,p)$	3.01	462	312.2 days
$^{46}Ti(n,p)$	4.06	124	83.83 days
$^{60}Ni(n,p)$	6.0	97.0	5.271 years
$^{63}Cu(n,\alpha)$	6.36	28.0	5.271 years

It is possible to avoid the procedure for finding the effective reaction cross-section in cases where a comparison of experimental and computational data is required. To do this, experimentally obtained reaction rates are compared with the calculated values, which are determined on the basis of equation (8) using group neutron flux densities and reaction cross-sections. However, as in the effective cross-section method, the problem again arises of selecting the neutron spectrum when obtaining the group cross-sections for the neutron activation detector cross-sections.

When using activation detectors, the directly measured quantity is the detector activity A_T, measured at a moment of time t, or the number of fissile detector fragments N_f recorded by track detectors. A simplified relation between the reaction rate and the detector activity has the form:

$$R = \frac{A_T \exp(\lambda t_h)}{N_{nuc.} \cdot k_o} \tag{10}$$

where

$N_{nuc.}$ is the number of nuclei in the target nuclide in the detector;

t_h is the detector holdup time and

k_o is the decay correction during irradiation.

In a steady-state neutron field, the decay correction will be

$$k_o = [1 - exp(-\lambda t_o)] \frac{P}{P_{nom.}} \tag{11}$$

where

t_o is the irradiation time at the power level P;

P_{nom} is the rated power level.

Where there is prolonged irradiation of detectors in a neutron field in a WWER, it is essential to know the reactor's operational history. Assuming that the neutron flux density at the point of irradiation is proportional to the reactor power, and presenting variations in power as a step function, the decay correction will take the form:

$$k_o = \sum_{n=1}^{N} \left[\frac{P_n}{P_{nom.}} exp(-\lambda t_{b,n}) (1 - exp(-\lambda t_{o,n})) \right], \tag{12}$$

where

N is the number of periods of operation at constant power;

P_n is the power during the n-th period;

$t_{o,n}$ is the operating time during the n-th period;

$t_{b,n}$ is the time from the end of the n-th period to the end of the N-th period.

For fissile nuclides the fission reaction rate will be:

$$R = \frac{N_f}{N_{nuc.} \varepsilon_f \cdot \sum_{n=1} \left[\frac{P_n}{P_{nom.}} t_{o,n} \right]} \tag{13}$$

where

E_f is the fission fragment recording efficiency.

Using the effective cross-section method, the integral neutron flux density and the reaction rate will be related by:

$$\varphi(>E) = \frac{R^E}{\sigma_{eff}^E} \tag{14}$$

where

$\varphi(>E)$ is the neutron flux density for neutrons of energy above E;

R^E, $\sigma^E_{eff.}$ is the reaction rate and the effective cross-section of a detector with an effective threshold E.

The neutron fluence is defined as:

$$\Phi(>E) = \varphi(>E) \cdot \sum_{n=1}^{N} \left[\frac{P_n}{P_{nom.}} \cdot t_{0,n} \right] . \qquad (15)$$

To derive the differential neutron spectrum using experimentally obtained reaction rates, various unfolding methods can be used, which are based on the solution of the system of equations (8). Problems of this type belong to the category of incorrectly set problems and require additional conditions or assumptions. Calculated neutron spectra used in pressure vessel dosimetry can serve as a priori information about the unknown spectrum. The algorithm for spectrum unfolding, which is based on the a priori spectrum, is executed by the SAND method. This and other unfolding methods and the spectra and cross-section libraries combined within the SAIPS system [5] are employed in the neutron dosimetry of pressure vessels.

It should be noted that the main elements of the experimental method of pressure vessel dosimetry that are responsible for introducing errors into the determination of integral and differential neutron characteristics are:

(1) The reaction cross-section for those interactions of neutrons and detector nuclei which lead to fission or the formation of radioactive nuclides;

(2) The selection of an a priori neutron spectrum to evaluate effective cross-sections or group reaction cross-sections;

(3) The dependence of the shape of the unfolded differential spectrum on the a priori information selected or on the assumptions made about the shape of the spectrum;

(4) The representation of local variations in neutron flux density at the point where the detector is located;

(5) The neutron flux gradient for the capsule containing the detectors;

(6) Corrections for burnup, neutron shielding by the capsule material, neutron depression in the detector material, photofission, etc.;

(7) Measurement of activity or track counting.

RESULTS OF WWER PRESSURE VESSEL NEUTRON DOSIMETRY

A great deal of attention has always been paid to the experimental determination of neutron characteristics in the WWER pressure vessel region. In order to obtain the most reliable and comprehensive data on neutron fields in individual standard WWER units, experimental channels have been installed. Table 3 lists reactors in which channels were mounted next to the pressure vessels [2, 6].

Table 3

Reactor	Pressure vessel radius, cm (inside-outside)	Channel positioning	Channel co-ordinates: Radius, cm	Height, cm	Angle in 30-degree sector
Novovoronezh-2		Vertical	176.5	-	17.5
WWER-365 Thermal output = 1320 MW	178-190	Vertical	192	-	17.5
Armenia-1 WWER-440	178-192	Vertical	239.8	-	10
Novovoronezh-5 WWER-1000	206.9-226.75	Vertical	293.5	-	30
South Ukrainian-1 WWER-1000	206.9-226.75	Azimuthal	237	96	-

On the basis of the results of activation measurements in the Novovoronezh-2 channels, the integral neutron spectrum in front of and behind the pressure vessel was determined in the 0.05-10 MeV energy range. To interpolate the neutron flux density in the 0.05 MeV region and to determine the effective reaction cross-sections more accurately, experiments were set up on a mock-up of the Novovoronezh-2 reactor pressure vessel [2]. The flux density of neutrons above 0.5 MeV in front of and behind the vessel, at a point level with the centre of the core was 1.72×10^{11} and 3.60×10^{10} $cm^{-2}s^{-1}$ respectively [2].

The validity of the calculation method of determining neutron characteristics in WWER pressure vessels using the DOT and ANISN programs has been confirmed by experimental data obtained from experimental channels in the Armenia-1, Novovoronezh-5 and South Ukrainian-1 units [6]. Table 4 compares calculated and experimental values for the integral neutron flux densities in the channels at a position level with the centre of the core (Armenia-1, Novovoronezh-5) and at the maximum flux density distribution (South Ukrainian-1). The authors of Ref. [6] concluded that the discrepancy between calculation and experiment did not exceed $\pm$ 20%. Experiments revealed that, along the length of the horizontal channel encircling the pressure vessel of the South Ukrainian-1 WWER-1000, the fast neutron flux density may differ by a factor of 2.

A unique experiment to determine the fast and thermal neutrons flux density in WWER-440 pressure vessels was performed at the Loviisa-1 plant [7]. For these measurements, metal samples scraped off from the inner surface of the reactor pressure vessel were used as neutron activation detectors. This was done after the reactor had been operating for three years. The integral fast neutron flux density distribution was determined from the reaction $^{54}Fe(n,p)^{54}Mn$ with a threshold energy of 2.8 MeV (Fig. 3). According to data from Ref. [7], the extrapolated value for the density of neutron fluxes above 1.0 MeV on the inner surface of the pressure vessel at a height of 110 cm was $(1.63 \pm 0.44) \times 10^{11}$ $cm^{-2}s^{-1}$. An analysis of the calculated and experimental data obtained in the space near the pressure vessel of WWER-440 reactors of the same type is presented in Ref. [8]. In Fig. 4, the experimental points have been inserted in the theoretical distribution of the density of neutron fluxes above 2.8 MeV, from which it can be seen that the correlation between calculation and experiment is satisfactory. However, it should be mentioned that no analysis was made of the neutron source distribution (corresponding to the calculated and experimental data in Fig. 4) in the cores of different WWER-440 reactors.

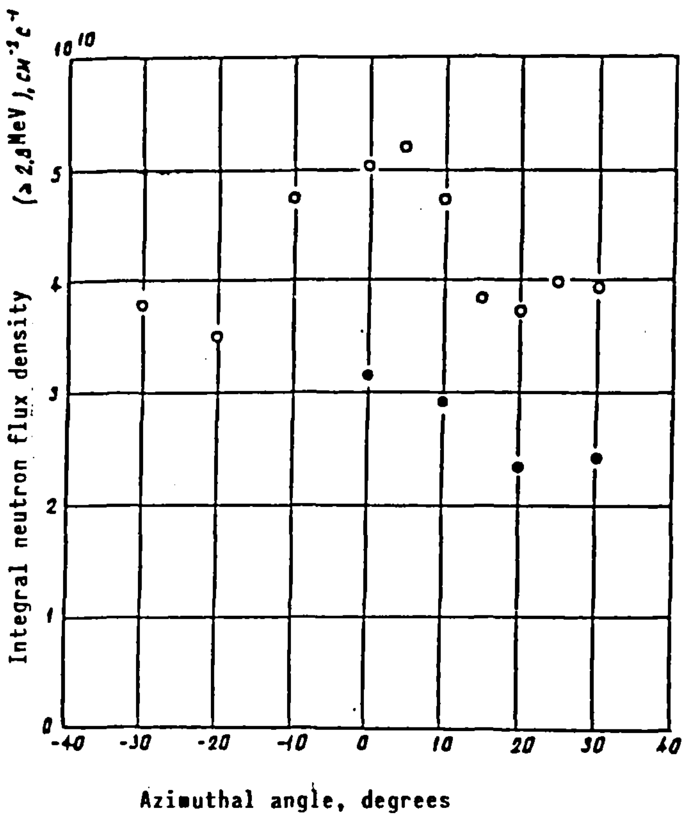

Fig.3:

Azimuthal distribution of the integral neutron flux density (≥2.8 MeV) on the inner surface of the pressure vessel of the Loviisa-I WWER-440 at a height of 110 cm (o) and 22 cm(•).

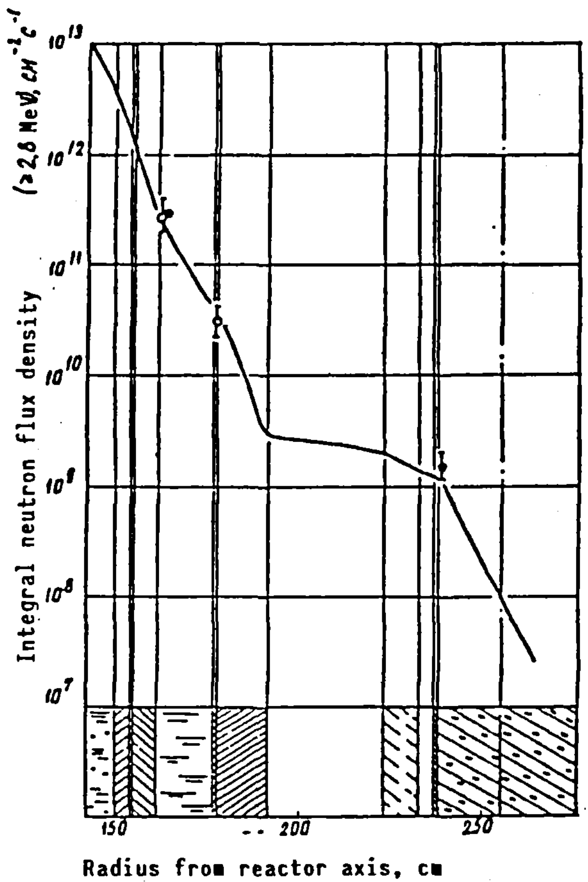

Fig.4:

Distribution of the integral neutron flux density (≥2.8 MeV) along the radius of a series-produced WWER-440:

• - experimental data from Armenia-I;

o - the experiment at Loviisa-I

—— - calculation

Table 4

Energy E, MeV	Neutron flux density (>E), $cm^{-2}s^{-1}$					
	Armenia-1		Novovoronezh-5		South Ukrainian-1	
	Experimental	Calculation / Experimental	Experimental	Calculation / Experimental	Experimental	Calculation / Experimental
1.0	$1.2 \cdot 10^{10}$	0.84	$5.9 \cdot 10^{8}$	0.96	$1.95 \cdot 10^{9}$	0.97
2.3	$2.28 \cdot 10^{9}$	0.92	$1.2 \cdot 10^{8}$	1.03	-	-
3.0	$1.65 \cdot 10^{9}$	0.79	$7.8 \cdot 10^{7}$	1.00	-	-
5.5	$3.4 \cdot 10^{8}$	1.06	$2.8 \cdot 10^{7}$	0.89	-	-
7.2	$1.69 \cdot 10^{8}$	0.89	$1.4 \cdot 10^{7}$	0.86	-	-

Table 5

Reactor	Type	Neutron flux density (E ≥ 0.5 MeV) on the inner surface of the pressure vessel, $cm^{-2} \cdot s^{-1}$	
Novovoronezh-2	WWER-365	$(1.72 \pm 0.22) \times 10^{11}$ [*]	[2]
Armenia-1	WWER-440	3.4×10^{11}	[6]
Novovoronezh-5	WWER-1000	3.5×10^{10}	[9]
South Ukrainian-1	WWER-1000	5.7×10^{10}	[6]

[*] The value presented is that for the experimental channel installed in front of the pressure vessel.

The published maximum values for neutron flux densities for energies above 0.5 MeV on the inner surface of WWER pressure vessels that have been published in the literature and can be used to define the degree of radiation embrittlement of pressure vessel steel are listed in Table 5. The data presented were obtained from the results of calculations tested and validated by separate measurements in experimental channels.

A great deal of experimental research has been carried out in recent years with the aim of validating and more accurately defining the neutron fluence incident on WWER pressure vessels. Measurements conducted by the method described in Ref. [10] without installing experimental channels in the gap have made it possible to obtain the azimuthal and height distributions of the neutron fluence and flux density in the pressure vessels of the main types of WWER and to study the variations in the absolute values of neutron characteristics in the course of a number of campaigns at the same units. The results obtained have demonstrated how effective out-of-vessel neutron dosimetry is in accurately defining and monitoring neutron characteristics in WWER pressure vessels. An analysis of the results of these experiments combined with neutron field calculations and research on control samples could solve the problems of neutron dosimetry and thereby contribute to the safety of pressurized power reactors.

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